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# Gas-cooled Nuclear Power Reactors

*Although the U.S. has only one such reactor, they have served well overseas. They have an attractive safety feature: a loss-of-coolant accident such as the one at Three Mile Island is all but impossible*

by Harold M. Agnew

In March, 1979, the nuclear power industry suffered a shock from which it has not yet recovered: the accident that disabled one of the nuclear reactors at Three Mile Island. It is ironic that an event that caused no discernible physical harm to anyone crippled the prospect for expanding nuclear power at the very time the nation was becoming generally aware of the need for new domestic sources of energy.

Although the experience at Three Mile Island demonstrated to the satisfaction of technically qualified people that present-day water-cooled nuclear reactors offer no significant threat to the health and safety of the general public, it also showed that such accidents and equipment failures can jeopardize the operability of the plant and place at risk the heavy capital investment it represents. In the extreme case an accident such as the one at Three Mile Island can threaten the financial survival of the operating utility.

Perhaps the principal lesson of Three Mile Island is that the current generation of nuclear power plants is vulnerable to certain rare events that can lead to a condition where the time available for responding correctly can be less than a minute. In such low-probability events if the appropriate actions are not undertaken immediately, the consequences can be extremely costly even when public safety is not at issue. It is reasonable to ask: Do we need to be content with nuclear reactors of a design such that operators must react correctly within a minute in order to prevent damage to the reactor? The answer is no.

That being the case, how did the U.S. nuclear power industry come to follow the path it did? The dominance in the U.S. of the light-water reactor has a simple explanation. The pressurized-light-water reactor is a straightforward adaptation of the highly compact reactor designed to propel the first nuclear-powered submarine, the U.S.S. *Nautilus*, launched in 1954. An electric-power version of the submarine reactor, built by the Westinghouse Electric Company, went into service at Shippingport, Pa., three years later. The General Electric Company soon introduced a reactor design of its own, the boiling-water reactor, in which the heat generated by nuclear fission was carried away from the core by steam rather than by pressurized hot water.

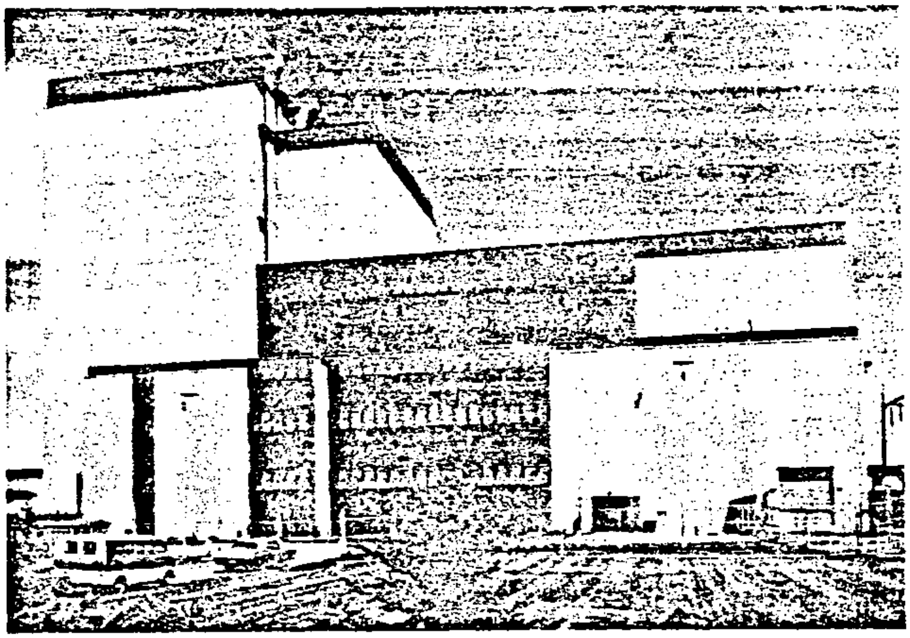
In both types of reactor it is essential that the reactor core not be uncovered, even briefly, lest the temperature in the core quickly rise and melt the metal jackets around the fuel pellets, as indeed probably happened at Three Mile Island. Light-water reactors are equipped with redundant safety features to cope with a "loss of coolant" accident. In such accidents the emergency equipment is designed to flood the core with water from a plentiful and assured source. When the normal coolant flow was interrupted at Three Mile Island, a sequence of improbable events, including apparent operator error, interrupted the delivery of the emergency cascade of water for too long a time.

All but one of the 71 commercially licensed and operating nuclear power plants in the U.S., which currently supply about 11 percent of the nation's electric power, are light-water reactors. The exception is a helium-cooled reactor, the Fort St. Vrain Nuclear Generating Station, which was accepted for service in the summer of 1979 by the Public Service Company of Colorado. The plant's rated capacity is 330 megawatts of electric power, or MWe, which is about a third the output of a standard commercial power plant. The reactor has been operating at up to 70 percent of rated power and has recently been released by the Nuclear Regulatory Commission for testing at up to full power.

The Fort St. Vrain demonstration plant was designed and built for the Public Service Company of Colorado by the General Atomic Company as a part of the Atomic Energy Commission's Power Reactor Demonstration Program. This followed the successful operation of a 40-MWe prototype, Peach Bottom Atomic Power Station No. 1, on the system of the Philadelphia Electric Company. In its seven and a half years of operation, from 1967 through 1974, the Peach Bottom reactor was available for service 86 percent of the time (except for scheduled shutdowns related to the research and development objectives of the reactor itself). The comparable figure for all U.S. nuclear reactors is about 66 percent.

The key safety features that differentiate the helium-cooled reactor from water-cooled reactors are two. First, since the reactor core is cooled by a circulating gas completely confined within a massive reactor vessel, the reactor cannot lose its primary coolant because of a rupture of pipes outside the vessel. Second, if the circulation of the gas is interrupted by some mishap to all of the main helium-circulation system, the temperature within the reactor core rises only slowly because the fuel elements are embedded in a massive matrix of graphite, which serves as the moderator for slowing down neutrons and which can absorb the heat released by fission products after the nuclear chain reaction itself has been halted. In a helium-cooled graphite-moderated reactor the nuclear reaction is halted by the insertion of control rods, similar to those in all reactors, or alternatively by the injection of small boron-containing balls that "poison" the reaction. In water-cooled reactors the loss of the coolant, which also acts as the moderator, stops the reaction.

If the emergency cooling systems in a light-water reactor should fail to function, the temperature in the reactor core would rise even after the reaction had been shut off because the fission products accumulated in the fuel elements would continue to release energy at a high rate. At the instant of shutdown, decay heat amounts to about 7 percent of the rated thermal output of the reactor, or about 210 megawatts in a water-cooled plant with a thermal rating of 3,000 megawatts (equivalent to an electric output of 1,000 MW). It is estimated that in such a loss-of-coolant accident the temperature of the cladding around the fuel elements would reach 3,000 degrees Fahrenheit and fuel failure would begin in as little as 50 seconds in a pressurized-water reactor and in less than two minutes in a boiling-water reactor. With a helium-cooled reactor, in a comparable event involving a system depressurization and the total failure of the helium-circulation system, more than an hour would be required for the temperature inside the core to reach 3,000 degrees F. At that temperature both the coated fuel particles and graphite fuel elements in a helium-cooled reactor would not be affected. The fuel particles and graphite can readily withstand temperatures of up to 4,000 degrees F., which would not be reached until at least 10 hours had elapsed. In short, there is ample time to institute a variety of reasoned emergency measures for restoring the flow of helium coolant.



**ONLY HTGR IN THE U.S. is near Denver, Colo. It is the Fort St. Vrain Nuclear Generating Station, designed by General Atomic and owned and operated by the Public Service Company of Colorado. The plant, which has a capacity of 330 MWe, was placed in operation three years ago and has since supplied more than two billion kilowatt-hours of electricity. On several occasions the forced circulation of coolant in the reactor core has been interrupted for periods of as much as 15 minutes without doing any detectable harm to the core or to the fuel elements.**

The virtues of gas-cooled graphite reactors have been widely recognized elsewhere in the world. In the 1950's and 1960's, when the U.S. had committed itself to light-water reactors, Britain and France developed gas-cooled graphite-moderated reactors, in which the coolant was carbon dioxide rather than helium. Britain now has more than 40 gas-cooled reactors in operation or under construction, France has seven and Italy, Spain and Japan have one each. More than 600 reactor-years of operating experience has been acquired with the European gas-cooled reactors. Such reactors have accounted for nearly a fifth of the total nuclear power generated in western Europe, Japan and the U.S. so far.

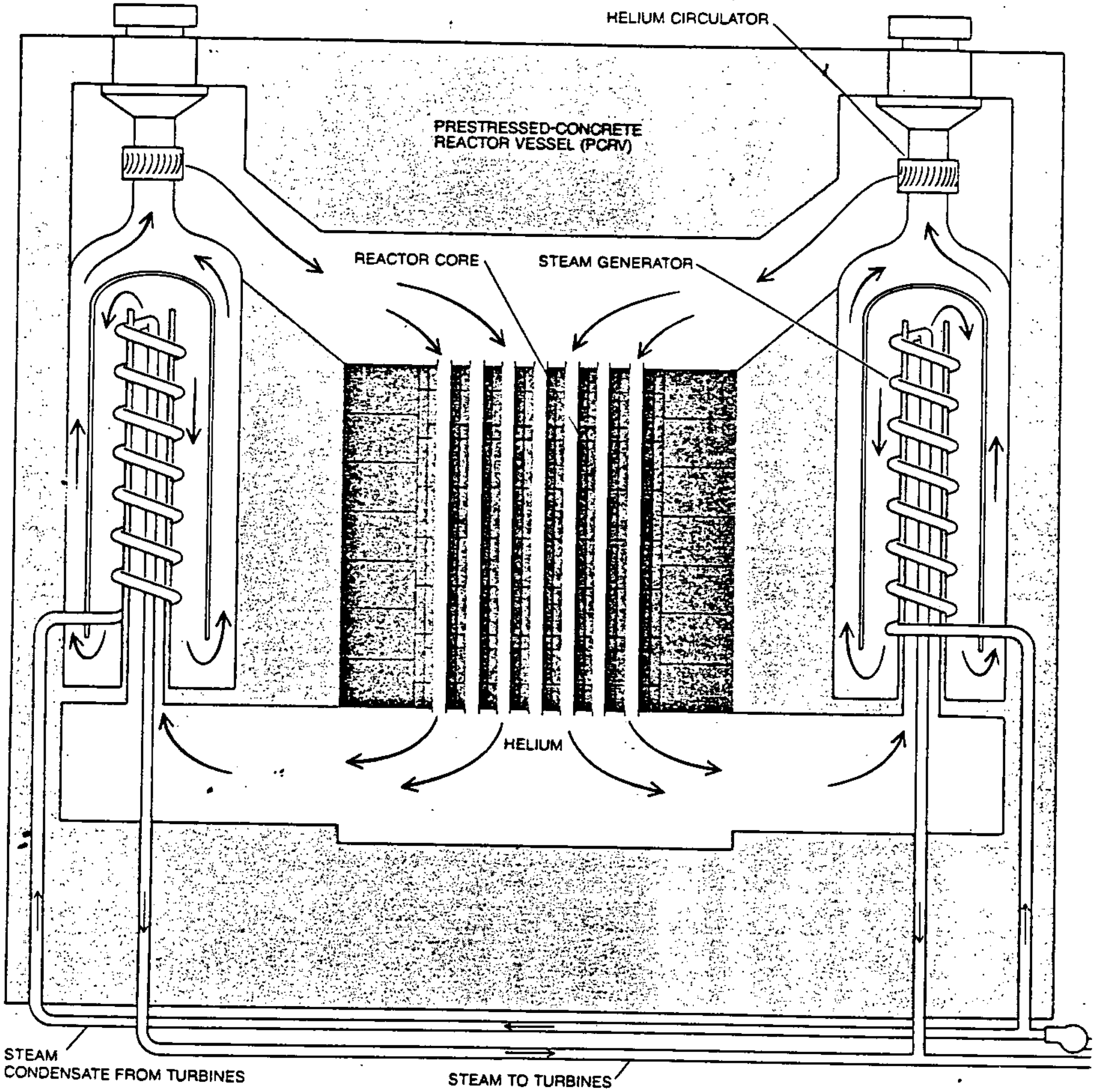
The British and French efforts were at an early stage in 1956 when a group of physicists, many of them with experience at the Los Alamos Scientific Laboratory, gathered at La Jolla, Calif., to consider the problem of designing a reactor that would be both more efficient and inherently more "forgiving" than the reactors then available. Among those present were H. A. Bethe of Cornell University, Freeman J. Dyson of the Institute for Advanced Study, Peter Fortescue of the Atomic Energy Research Establishment at Harwell in England and Frederic de Hoffmann, who was then president of General Atomic. Out of these early deliberations, aided by concepts from Britain and France, evolved the concept of the high-temperature gas-cooled reactor, or HTGR, tested at Peach Bottom and on a larger scale at Fort St. Vrain. Because the U.S. has plentiful supplies of helium, that gas could be selected as a coolant instead of carbon dioxide. Helium has the important advantage that it is stable to the high radiation flux in the reactor, does not become radioactive, is chemically inert and has excellent heat-transfer characteristics.

The attractive features of HTGR's were summarized by Joseph M. Hendrie, chairman of the Nuclear Regulatory Commission, in testimony before a congressional subcommittee in March, 1980. Such reactors, he said, "have efficiencies as good as the best fossil-fuel plants and are substantially more efficient than the water-cooled reactors. They not only get better thermal efficiency but also get better energy utilization out of each pound of uranium that is mined, better, in fact, by probably 15 or 20 percent than the best estimates for advanced light-water fuels." He added that HTGR's "have some safety advantages. They are machines in which you don't have to do a lot of things in a hurry if something goes wrong because the core structure is a great massive pile of graphite, a very high-temperature and stable material, so that if you get a power dropoff or the plant circulators go out, [you have time] to sit down and think about what to do."

The first series of gas-cooled reactors built in Britain were called Magnox reactors because the fuel rods, which contained natural unenriched uranium, were clad in a magnesium alloy. The reactor core, incorporating many tons of graphite, was housed in a large and expensive steel pressure vessel many times bigger than the pressure vessels needed for light-water reactors. Then in 1958 French engineers showed that the steel vessel could be replaced with a vessel of prestressed concrete that could be constructed in sizes large enough to house the entire reactor system, including the steam generators. The prestressed-concrete reactor vessel, or PCRV, is kept in compression at all times by a network of redundant, tensioned steel tendons that can be monitored and retensioned or even replaced if necessary. Tightness against leaks is ensured by a steel liner affixed to the inside of the PCRV, which acts only as a membrane seal to contain the coolant. The liner and the walls of the PCRV are cooled by water circulating through tubes that are welded to the outer surface of the vessel.

PCRV's were subsequently adopted for all French and British gas-reactor systems. The high degree of safety afforded by the concrete vessel contributed to the British decision to construct a second generation of reactors known as advanced gas reactors (AGR's) near urban sites. In this second generation the fuel was uranium oxide, a ceramic, clad in stainless steel, a change made possible by the adoption of slightly enriched uranium. With the new fuel AGR's could operate at higher temperatures than the Magnox-fueled reactors and were able to "burn" more of the uranium 235 in the fuel before refueling became necessary. With higher temperatures the efficiency of electric-power generation was raised from about 30 percent to a little more than 40 percent.

In the U.S. the Atomic Energy Commission (a predecessor agency of the Department of Energy) nurtured interest in gas-cooled reactors in the 1950's and 1960's by supporting the study of several advanced reactor concepts. One of the AEC's main objectives was to reduce the amount of uranium required




**REACTOR CORE AND STEAM GENERATORS of the HTGR are enclosed in a massive prestressed-concrete reactor vessel (PCRV). For a reactor designed to generate 860 MWe the PCRV would be 102 feet in diameter and 95 feet high. (A pressurized-light-water reactor of slightly larger capacity is shown at the same scale in the illustration on the opposite page.) The graphite core of the 860-MWe HTGR fills a cylindrical volume 26 feet in diameter and 21 feet high. Helium at a pressure of 1,050 pounds per square inch is circulated through some 27,000 vertical channels in the core by four primary circulators, of which only two appear in this cross section. The helium emerges from the reactor core at a temperature of 1,266 degrees Fahrenheit and enters the base of the steam generators, where it makes two passes over an array of helical and straight steam coils. Water boils upward through the coils and is further heated as it passes downward to emerge as superheated steam with a temperature of 1,000 degrees F. and a pressure of 2,500 pounds per square inch. Not shown are three coolant loops in which water-cooled heat exchangers can remove heat from circulating helium when the steam-generating loops are out of service. After a reactor shutdown fission products in the core release heat at a rate that is high initially but declines exponentially.**

per unit of electric power; at that time uranium resources appeared scarce in relation to the projected needs. As a result the study emphasized reactor concepts that were either breeders or advanced converters. A breeder creates at least one atom of new fuel for each atom of fuel consumed. Advanced converters generally create from .7 to one atom of fuel for each atom consumed. Light-water reactors yield between .5 and .6 atom of fuel for each atom consumed. The high-temperature gas-cooled graphite-moderated reactor qualifies as an advanced converter. It was one of the designs that survived the inevitable weeding out. The HTGR had strong support from the utility industry because it is competitive in capital costs with light-water reactors and because it exploits a uranium-thorium fuel cycle with a low uranium consumption and therefore low fuel costs.

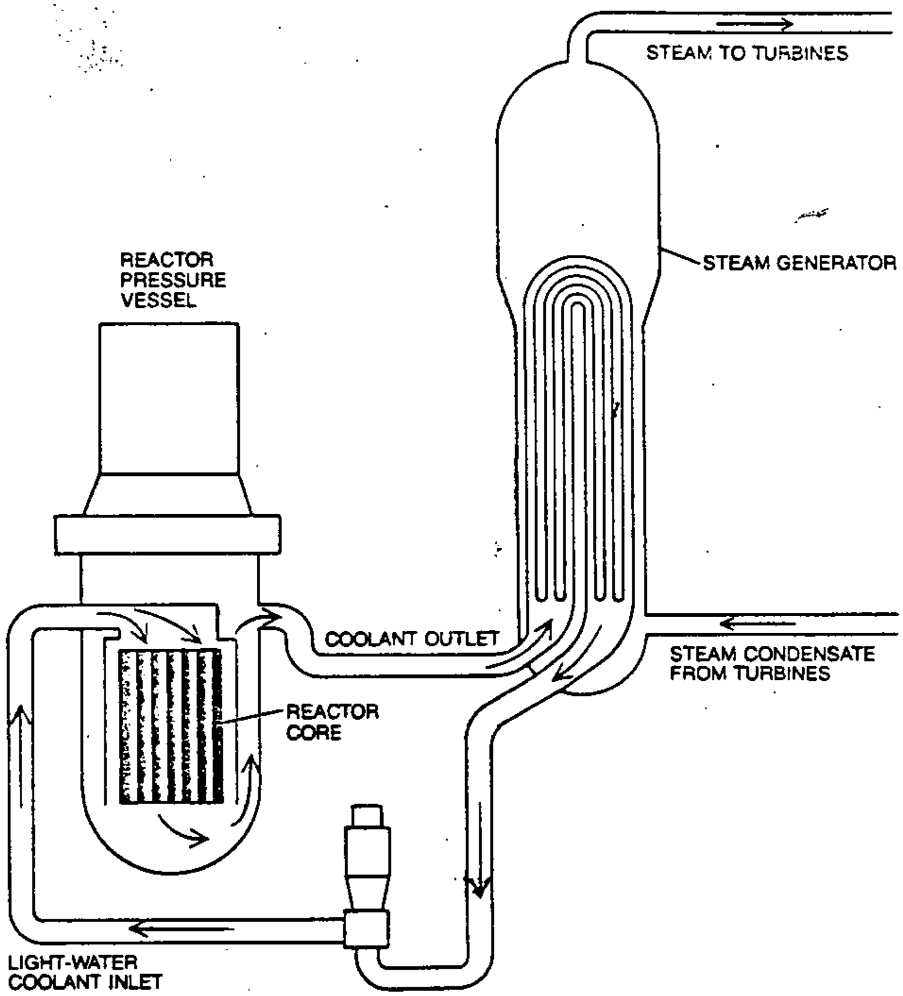
The continuing evolution of gas-reactor technology in Europe and the U.S. has led to a convergence in at least two important particulars for the next stage in the development of gas-cooled reactors. Helium replaces carbon dioxide as a coolant and the reactor core is charged with nuclear fuel in a unique system that dispenses with the need for a metal cladding. The two features have been demonstrated not only at Peach Bottom and Fort St. Vrain but also in two European reactors. The British operated a helium-cooled 20-MW thermal test reactor in southern England from 1965 to 1976. In Germany an HTGR of 15 MWe (called the AVR) has been generating electric power since 1967, with the outlet gas temperature being as high as 950 degrees Celsius. (The temperature of water leaving the core of a pressurized-light-water reactor is about 610 degrees F., or 321 degrees C.) A 300-MWe plant based on the AVR experience is now under construction in Germany and is scheduled for start-up in 1984 or 1985. In the U.S. the Fort St. Vrain reactor of 330 MWe has provided more than two billion kilowatt-hours of power since 1978 and has demonstrated the fuel performance and safety characteristics of a contemporary HTGR design. The reactor has been subjected to test transients up to and including the complete loss of forced-coolant circulation with no adverse effects on the reactor core or on other primary components of the system.

On the basis of the Fort St. Vrain experience General Atomic, in cooperation with Gas-Cooled Reactor Associates (an organization of U.S. utility companies) and the Department of Energy, has developed a reference design for an HTGR of 860 MWe. The goal has been a design that is simple and conservative and that places high emphasis on the safety and protection of capital investment. The reactor core is contained within a multicavity prestressed-concrete reactor vessel. Helium leaves the core at 1,266 degrees F. (reduced from 1,494 degrees F. at Fort St. Vrain) and passes through four primary coolant loops, where steam is generated at a temperature of 1,000 degrees F. and a pressure of 2,500 pounds per square inch.

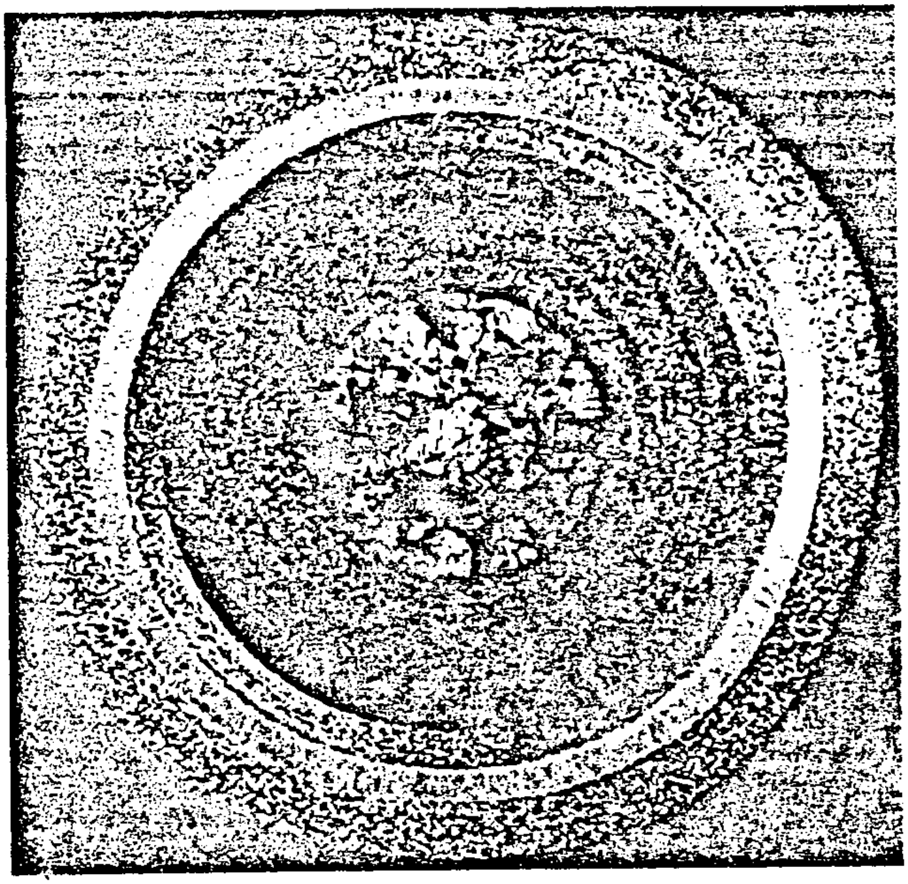
Helium is forced through each coolant loop by a circulator driven by an electric motor. (The Fort St. Vrain circulators are driven by steam.) The core also is provided with an auxiliary cooling system consisting of three loops, each sufficient by itself to deliver 100 percent of the required cooling when the helium in the reactor vessel is at the normal working pressure of 1,050 pounds per square inch, or 50 percent of the cooling when the vessel is depressurized. The helium that passes through the auxiliary cooling system is cooled with water circulated by electrically driven pumps that can be powered, if need be, by diesel generators.

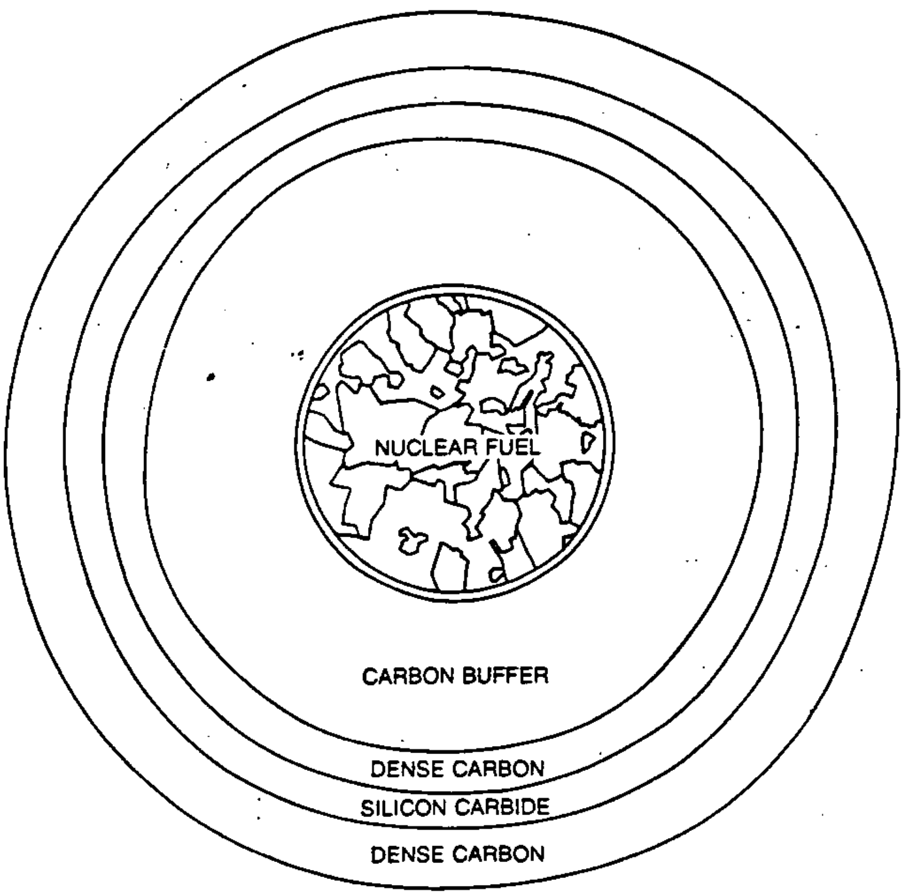
The combination of a stable, inert gas for a reactor coolant and a highly temperature-resistant graphite core structure allows steam to be generated at the high temperatures and pressures found in the modern electric-power plants that burn fossil fuel. The net electric-generating efficiency of the HTGR reference design is 38.5 percent, slightly below the 39.2 percent achieved at Fort St. Vrain. The small reduction was made in the interest of simplifying the steam-generating system and to furnish still further operating and safety margins.

A fundamental property of the helium coolant, a confined gas that cannot possibly condense to liquid form in the system, is that it follows a linear temperature-pressure relation; therefore instru-




**PRESSURIZED-LIGHT-WATER REACTOR** has been the commonest type of nuclear power reactor in the U.S., with 44 reactors now in operation. Another 24 reactors are of the boiling-water type, in which heat is carried off from the core by steam rather than by pressurized heated water. The core of a pressurized-light-water reactor rated at 1,100 MWe is shown here. It is housed in a steel pressure vessel about 15 feet in diameter, 40 feet high and from six to 11 inches thick; the vessel is designed to operate with an internal pressure of 2,250 pounds per square inch. The coolant water leaves the reactor at 610 degrees F. and passes to four steam-generating loops, only one of which is shown here. Steam emerges from the generator at 540 degrees F. and a pressure of 1,000 pounds. At this temperature and pressure the system's thermal efficiency is only 32 to 33 percent, compared with 38.5 percent for an HTGR system.






**FUEL PARTICLE developed for HTGR systems is .03 inch in diameter, about the size of a grain of sand. A cross section of the particle is enlarged 150 diameters at the top. The nuclear fuel itself is the crystalline-like material in the center. It consists of uranium oxycarbide in which for best performance the content of the fissionable isotope uranium 235 is enriched to 93 percent. Layers of carbon and silicon carbide are built up by a high-temperature process.**

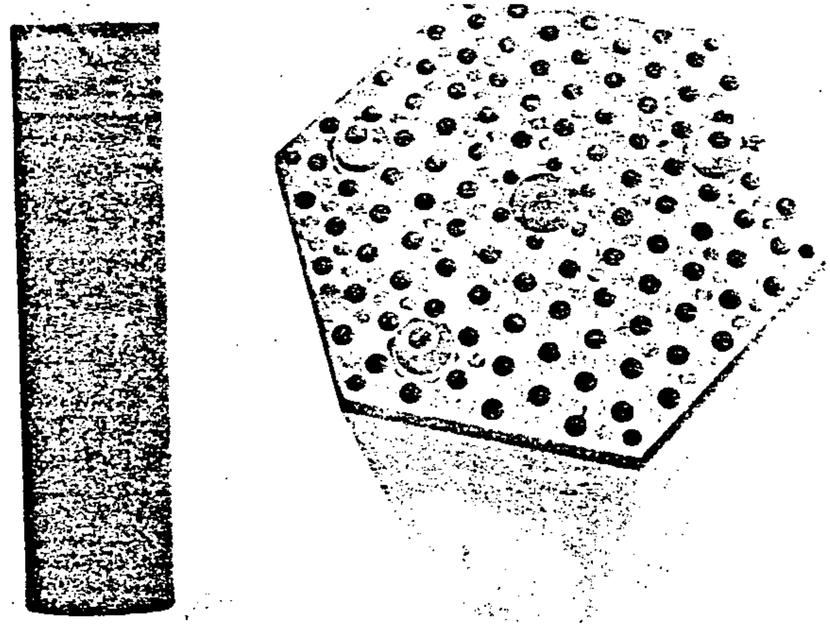
ment readings of temperature and pressure can provide independent checks on each other. Because there is no liquid-gas interface, as there is in boiling-water reactors (and in pressurized-water reactors under certain emergency conditions), a single unambiguous signal—pressure—always indicates the presence and physical condition of the coolant. Rapid depressurization of the primary cooling system can be tolerated without concern that voids have formed and left part of the core uncovered, as can happen when pressure is released from water that is above its atmospheric boiling point.

The Fort St. Vrain experience has verified several important safety and operating advantages of HTGR's. Operating and maintenance personnel have received exposures to radiation far below the limits established for nuclear plants. Fewer than 10 workers out of a total of several hundred have received amounts of radiation that were even measurable.

The Fort St. Vrain system has responded smoothly and gracefully to load changes caused either by transient excursions in the power-generating cycle or by the temporary shutoff of equipment within the plant. Because the core of the HTGR is large and releases less heat per unit volume than light-water reactors do and because the massive core, incorporating some 1,500 tons of graphite, has a large capacity to absorb heat if coolant flow is reduced or interrupted, the reactor responds slowly to an unexpected operational upset, allowing the operators enough time to take appropriate action: hours rather than seconds.

At Fort St. Vrain five such upsets have interrupted the forced circulation of helium for extended periods without giving rise to a measurable increase in the temperature of the core or harming the plant or the fuel in any way. The risk of damaging the reactor or the reactor core through operator error is virtually eliminated. Thanks to the HTGR's thermal stability the system for bringing the activity of the reactor to a halt by the insertion of neutron absorbers and the systems for emergency cooling can be of simple design. There is also ample time for such systems to be actuated manually if it is allowed by regulation. One consequence of the Three Mile Island accident is that the Nuclear Regulatory Commission now requires the full-time presence of an on-site expert, called a shift technical adviser, at nuclear power plants. Fort St. Vrain is the only reactor exempted from this rule; an expert is not required to remain on the site but is on call to report within an hour.

The prestressed-concrete reactor vessel is incorporated in the design as a major safety feature. First, a catastrophic rupture of the PCRV is such a remote possibility that risk analysts characterize it as being ... don that gives the PCRV its strength is independent and redundant; the vessel is in a constant state of compression. Second, the PCRV is designed to withstand an ultimate pressure of more than twice the normal operating pressure, or some 2,400 pounds per square inch. Any crack in the steel liner that might result from excessive pressure can do no more than give rise to a slow gas leak; such leaks tend to seal themselves when the pressure is reduced slightly. Third, total depressurization can result only if there is a failure of one of the pipe penetrations or small service lines that pass through the wall of the PCRV. Such a hypothetical failure is an extremely low-probability event. Moreover, at each penetration site the vessel is equipped with flow limiters that prevent the rapid release of gas that could cause structural damage to the core or to the cooling system.



**FUEL ROD AND FUEL BLOCK** for an HTGR are shown respectively at the left and the right. The fuel rod, about 2.5 inches in length, consists of tens of thousands of fuel particles bound in a graphite matrix. Each fuel block, which is approximately 14 inches across and 31 inches high, holds 1,656 fuel rods packed in hexagonal arrays. The numerous empty channels in the block are paths for the flow of helium. The large central hole accommodates a mechanism for inserting the fuel blocks in the core of the reactor. The core of an 860-MWe reactor will require 3,512 blocks. Each 270-pound fuel block contains on the average 1.54 pounds of U-235 and 35 pounds of thorium 232. In its four-year residence in the reactor such a block would yield energy equivalent to 2,500 tons of coal or 12,000 barrels of fuel oil. If the unburned U-235 and the U-233 created from thorium were recovered and recycled, the energy equivalent of the original nuclear fuel would rise to some 11,000 tons of coal or 54,000 barrels of oil.

The improved performance characteristics of the HTGR also offer several environmental advantages over the current generation of reactors. Because an HTGR operates at an efficiency of about 39 percent compared with an efficiency of about 33 percent for light-water reactors, an HTGR releases about 25 percent less waste heat to be dissipated into the surrounding environment. If the heat, in the form of hot water, is rejected into a nearby lake or river, concern about raising temperatures to a point harmful to the aquatic ecosystem is reduced proportionately. If cooling towers are used to dissipate the heat, they consume less water and can be smaller and less expensive. If cooling ponds are used, an HTGR plant with about a third more megawatts of capacity than a light-water plant can be sited on a pond of a given size without exceeding a specified pond temperature. Where dry cooling towers must be adopted to meet environmental regulations or fit available water supplies, the loss of plant capacity in hot weather will be only about half as great with an HTGR as it is with existing nuclear power plants. As a result an HTGR plant can be situated at a remote arid or semiarid site with a smaller penalty in cost.

The level of radioactivity in normal discharges from all nuclear power plants is carefully monitored. An HTGR plant inherently releases into the plant process streams less radioactivity and at lower concentrations than a light-water reactor does. In addition an HTGR incorporates features that will ensure that releases of radioactivity from the plant to the environment are essentially zero. Routine decontamination procedures can be expected to produce small volumes of low-level liquid wastes (less than 2,000 gallons per year with a total activity of less than 150 curies). Such small volumes can be shipped off-site with little difficulty or retained on-site. The solid wastes produced by an HTGR should total less than 2,000 cubic feet per year. Some 80 percent will consist of low-level waste (such as paper, filter elements and spent resins) that is only slightly contaminated and can be shipped off-site in drums for burial or burning with virtually no effect on the environment. The remaining 20 percent will be intermediate-level waste, consisting chiefly of reflector blocks, which must be periodically replaced. Such waste can be shipped off-site in shielded 55-gallon transport casks for long-term safe disposal.

Helium-purification and gas-recovery systems incorporated in the standard HTGR plant should reduce the radioactive levels in released gases to several orders of magnitude below the current Government regulation of five millirems per year. Tritium (the radioactive isotope of hydrogen) generated within the primary system of an HTGR is removed in the helium-purification system by an oxidizer that converts the tritium into tritiated water, which is subsequently solidified and handled as solid waste that can be readily isolated as the tritium decays. (The half-life of tritium is 12.26 years.)

The HTGR has evolved a number of features that simplify operation, maintenance and refueling. For example, the entire primary coolant, helium, is confined within the prestressed-concrete reactor vessel. The PCRV itself provides all the necessary shielding for personnel, so that maintenance work can be done throughout the reactor building while the plant is in operation. Because the entire secondary steam system is essentially free of radioactivity all equipment in the steam cycle outside the PCRV, including the turbine-plant equipment, can be operated and maintained as it would be in a plant fired with fossil fuel. Because the amount of steam flowing to the turbines in an HTGR plant is only about 60 percent as large as that flowing to the turbines in a light-water power plant of the same output, all the equipment associated with the steam and feed-water cycles of an HTGR plant is small and therefore easier to maintain. In general, maintenance, repair and handling costs are lower in an HTGR plant than they are in light-water plants because helium, unlike water, is inert, nonradioactive and noncorrosive.

One big advantage of gas as a coolant is its transparency, which makes it possible to inspect many areas within the PCRV visually. The radiation shielding inherent in the design of the PCRV makes it possible to carry out many inspection and maintenance tasks while



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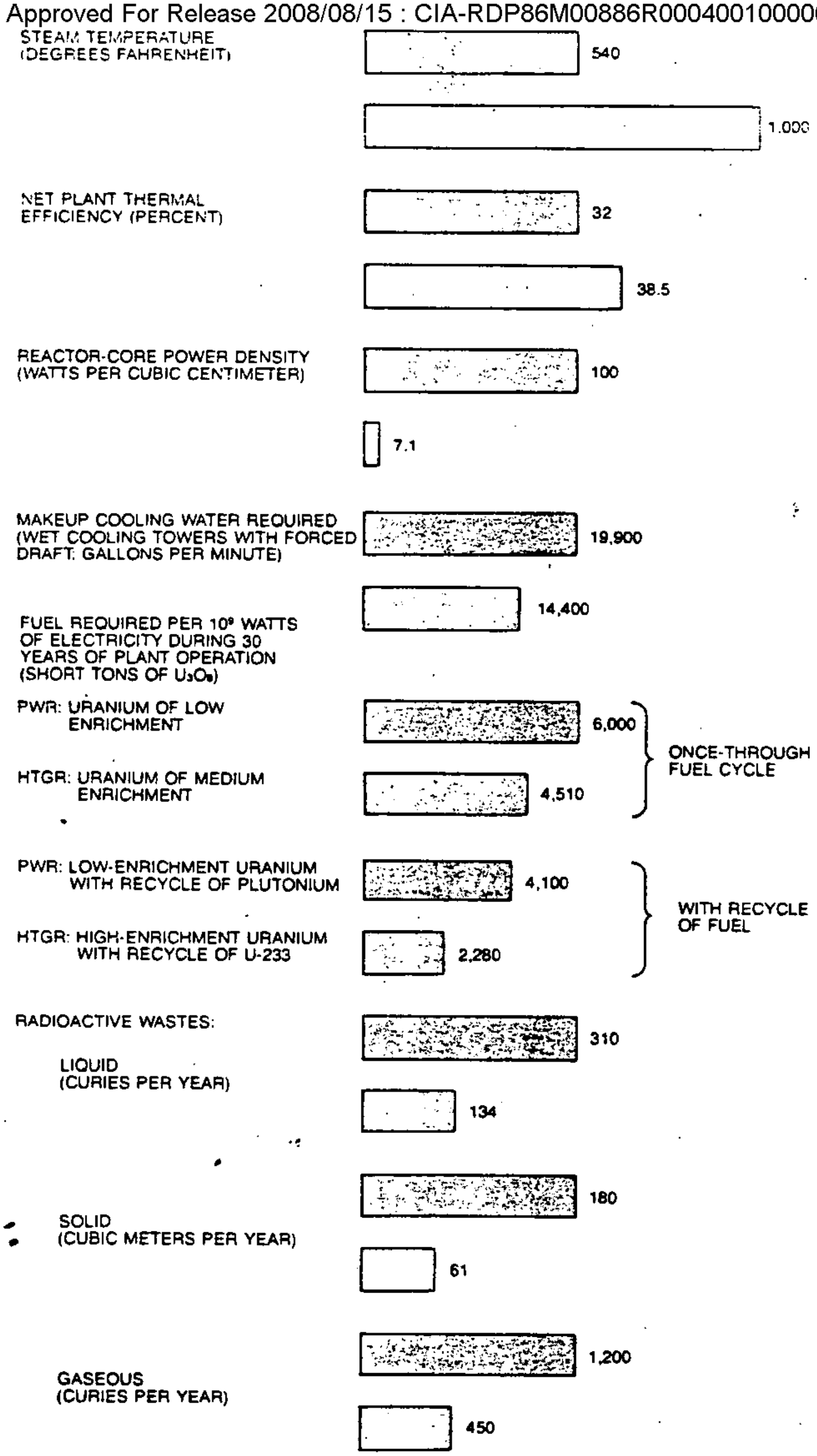



**OPERATING CHARACTERISTICS of the 860-MWe HTGR (*color*) are compared with those of a pressurized-water reactor of the same generating capacity (*gray*). The lower fuel consumption of the HTGR can be attributed in part to higher thermal efficiency and in part to the fact that for each atom of U-235 consumed in the HTGR about .7 atom of new fuel is created. The pressurized-water reactor creates less than .5 atom of new fuel for each atom consumed. With a once-through fuel cycle both systems convert a certain fraction of U-238 or Th-232 atoms into isotopes of plutonium or uranium, some of which are beneficially consumed before the fuel needs replacing. If the spent fuel could be recycled (which was contrary to the policy of the last Administration), it would be preferable to fuel an HTGR with a mixture of highly enriched uranium (about 93 percent U-235) and thorium. Some of the thorium would be converted into fissionable U-233, which could be recovered and recycled to replace U-235 in subsequent fuel charges. Smaller volume of radioactive wastes from an HTGR results partly from its higher efficiency and partly from advantages of helium over water as a coolant.**

the reactor is running, which reduces the time the reactor must be taken out of service for such purposes.

Essentially all structural members of the PCRV, such as the vertical tendons and the circumferential cable wrapping, can be inspected visually while the reactor is operating. Selected members are continuously monitored for changes in tension or strain that would indicate a deterioration in performance. If necessary, any structural member can be replaced. All external concrete surfaces, except those immediately surrounding the ports for the control rods, can be inspected visually while the plant is running. The control-rod ports and the surfaces surrounding the site where the control-rod drives penetrate the PCRV can be readily inspected in the course of refueling.

Recent refueling experience at Fort St. Vrain has demonstrated the ease of handling the HTGR's block-type fuel elements. About 240, or a sixth, of the fuel elements were removed from the core and replaced with fresh fuel; the other 1,240 elements were left in place. The refueling crew was exposed to such low levels of radiation that measuring them called for a microrem meter. By extrapolating from existing data one can calculate that the sum of the integrated man-rem exposure for the entire refueling operation following on the operation of the reactor at full power will be less than five man-rem. Federal regulations currently limit individual workers to five rem over a period of a year.

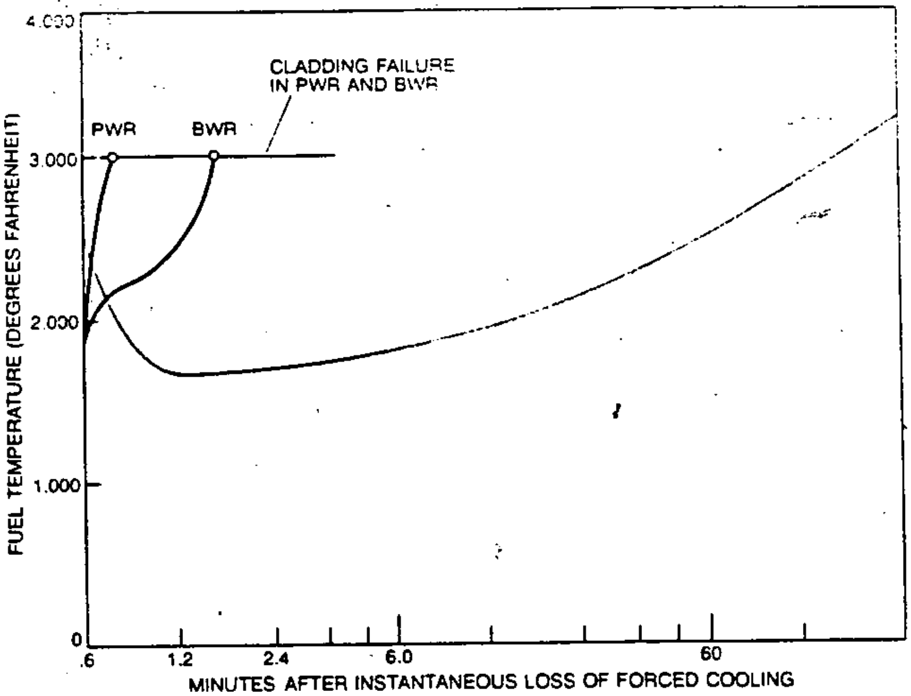
Each HTGR fuel element is a graphite block, hexagonal in cross section, 14 inches wide and 31 inches long. The block is perforated lengthwise with 72 coolant channels and 138 blind holes for fuel. Graphite is an ideal choice as a moderator and a structural material because its strength actually increases with temperature. In the reference design the graphite fuel blocks are stacked in columns of eight. This axially segmented arrangement facilitates fabrication, handling and refueling.

The convenient block configuration has been made possible by the development of a specially coated fuel particle. The kernel of each particle is a microsphere of uranium oxycarbide (suitably enriched in uranium 235) about .01 inch in diameter. Around each kernel thin layers of carbon, pyrolytic carbon and silicon carbide are applied at high temperature, yielding a tightly encased particle with a total diameter of about .03 inch. A similar form of encapsulation is used for the thorium particles. The technique ensures the containment of the fission products. The tiny spheres are tested in batches of 2,000 for structural integrity when they are exposed to a radiation flux that simulates the internal environment of the reactor. The particle-production process, which is semiautomatic, and the rigorous testing procedure work together to achieve a close control of quality.

Although severe and unforeseen service conditions in one region of the reactor core might cause the particle coating to fail and release fission products, the failure would be limited to the area directly involved. In most reactors, where the cladding of the fuel elements runs the entire length of the reactor core, an operating upset that ruptures a small section of cladding could release fission products from the entire length of the fuel rod. The performance of the fuel elements at Fort St. Vrain has fully met design expectations. Indeed, the release of fission products has been well below the predicted levels. In sum, the fission-product barriers in the HTGR fuel element have been demonstrated to have a high degree of reliability.

The properties of the HTGR make it possible to exploit a wide variety of nuclear fuel cycles with it. The cycle that has been most intensively studied and tested is the uranium-thorium one, in which fully enriched uranium (93 percent U-235) serves as the primary fissile material and thorium (Th-232) serves as a "fertile" material. In the reactor the thorium absorbs neutrons and is ultimately converted into the fissile isotope uranium 233, which can be recycled in subsequent fuel reloadings. The Fort St. Vrain reactor is fueled with uranium enriched to 93.5 percent U-235, in combination with thorium. The design of the plant allows the use of either fully enriched or medium-enrichment uranium (about 20 percent U-235). The HTGR fuel-cycle costs, under the current restraints on fuel reprocessing and recycling, are essentially equivalent to those of other commercial plants. Unless the policy is changed by the Administration spent fuel is to be stored indefinitely, without the recovery either of the unspent U-235 or of the U-233 or plutonium created during the operation of the reactor. This fuel cycle is commonly called the stowaway cycle.

If an HTGR were operated on a stowaway uranium-thorium cycle with fully enriched uranium, it would consume about 20 percent less uranium over its 40-year life than a light-water reactor would. If both types of reactor could be operated with a full recovery of their uranium and plutonium, the HTGR would consume about 50 percent less uranium. The HTGR therefore offers the opportunity of saving substantial amounts of uranium with either a stowaway policy or a full-recycle one, provided the reactor is designed to accept fully enriched fuel. The significance of the potential uranium saving can be appreciated when one considers that the total fuel cost over the life of a nuclear power plant is roughly equal to the total initial cost of the plant.

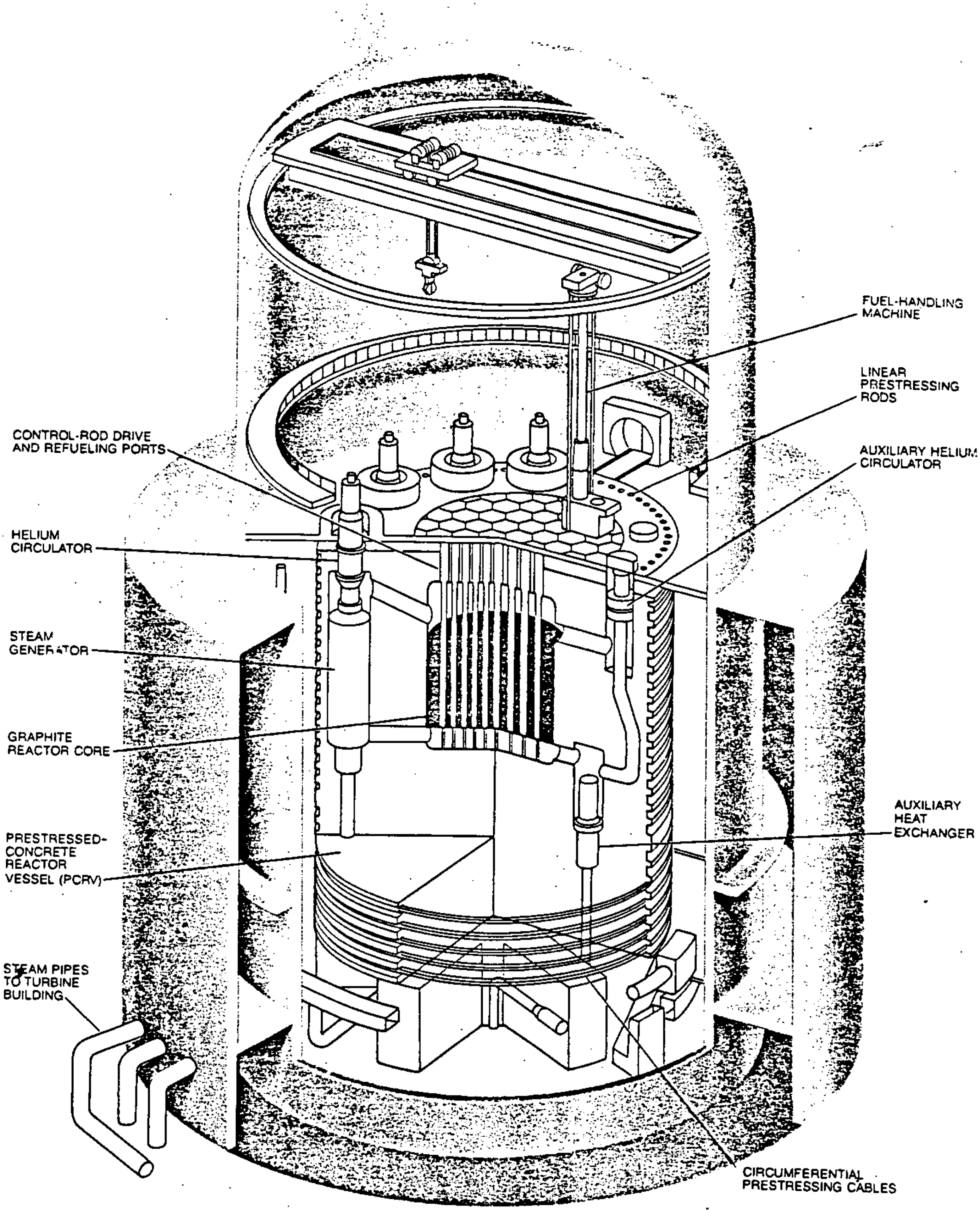



**INHERENT SAFETY OF AN HTGR is shown in graphs that compare the temperature in the core of an HTGR, of a pressurized-water reactor and of a boiling-water reactor following a hypothetical loss-of-coolant or loss-of-forced-circulation accident. In the water-cooled reactors the nuclear reaction is halted automatically by the loss of water, which serves as a moderator. In the HTGR the reaction must be stopped by the insertion of control rods that absorb neutrons. At the moment of shutdown decaying fission products in the fuel release heat at a rate equivalent to 7 percent of the thermal output of the reactor. The heat release falls to 1 percent in about two hours and to .5 percent in 24 hours. In the water-cooled reactors, in the absence of emergency cooling, the temperature of the cladding of the fuel would rise in less than two minutes to 3,000 degrees F., causing the cladding to fail. In the HTGR the mass of the graphite moderator would absorb the heat released by fission products, so that 3,000 degrees would not be reached for at least an hour. A temperature high enough to damage the graphite core (about 4,000 degrees) would be attained only after at least 10 hours without forced cooling.**

Over the past six years orders for about 55 nuclear power plants have been canceled. Only six years ago U.S. utilities had demonstrated interest in constructing 10 HTGR plants. Once the Fort St. Vrain reactor has been brought up to full power, which is scheduled for this summer, and has demonstrated the exceptional safety and reliability that its designers confidently predict, it is reasonable to assume that U.S. utilities will look favorably on the HTGR when they are again ready to place orders for nuclear power plants.

| YEAR | AVAILABILITY OF NUCLEAR STEAM SUPPLY SYSTEM (PERCENT) | PLANT AVAILABILITY (PERCENT) | OVERALL PLANT CAPACITY FACTOR (PERCENT) |
|---|---|---|---|
| 1967 | 81 | 78 | 69 |
| 1968 | 88 | 88 | 82 |
| 1969 | 86 | 84 | 67 |
| 1970 | 95 | 95 | 88 |
| 1971 | 90 | 87 | 78 |
| 1972 | 71 | 71 | 58 |
| 1973 | 95 | 94 | 78 |
| 1974 | 96 | 95 | 70 |
| CORE 1 AVERAGE | 85 | 83 | 73 |
| CORE 2 AVERAGE | 89 | 88 | 74 |
| TOTAL LIFETIME STATION AVERAGE | 88 | 86 | 74 |

**RELIABILITY OF FIRST HTGR PLANT designed by General Atomic, the Peach Bottom Atomic Power Station No. 1, is attested to by the statistics shown here. Apart from scheduled down time or time lost for reasons unrelated to the reactor, the HTGR was available for supplying steam for power generation 88 percent of the time. In achieving 74 percent of its rated electric-generating capacity over its seven-and-a-half-year lifetime the Peach Bottom reactor exceeded the typical figure of 66 percent achieved by light-water reactors operated by utilities.**




**HIGH-TEMPERATURE GAS-COOLED REACTOR (HTGR)** is a second-generation system more efficient than the 71 light-water power reactors that now supply about 11 percent of U.S. electricity. In this HTGR designed by the General Atomic Company the moderator (the material that slows neutrons in the reactor core) is graphite and the coolant is helium. In light-water reactors ordinary (but demineralized and conditioned) water serves both as the moderator and as the coolant. The HTGR shown would have an output of 860 megawatts of electricity (MWe), slightly less than that of the largest power plants, which generate 1,000 MWe ... mal efficiency of 38.5 percent, which is comparable to the efficiency of the best fossil-fuel plants and is higher than the 32 to 33 percent attained by current light-water reactors. Because the core of the HTGR contains nearly 1,500 tons of graphite, which has a high capacity for absorbing heat, an HTGR is much less likely to be damaged than a light-water reactor if there is an interruption in the flow of coolant or a loss of coolant. It was such an interruption that caused the accident at the Three Mile Island nuclear power station near Harrisburg, Pa. The reactor core and steam-generating system of the HTGR are ...te vessel with walls some 15 feet thick.

# THE AUTHORS

HAROLD M. AGNEW ("Gas-cooled Nuclear Power Reactors") is president of the General Atomic Company, which he joined in 1979 after more than eight years as director of the Los Alamos Scientific Laboratory. His work on the development of nuclear energy dates from the early 1940's, when as a recent graduate of the University of Denver he joined the small group that worked with Enrico Fermi on the first nuclear-fission chain reaction. In 1943 Agnew joined the Los Alamos laboratory to participate in the development of the atomic bomb. From 1946 to 1949 he was again with Fermi at the University of Chicago, obtaining his Ph.D. there in 1949. Thereafter Agnew was at Los Alamos except for three years (1961–64) as scientific adviser to the Supreme Allied Commander in Europe.

STEVEN WEINBERG ("The Decay of the Proton") is Higgins Professor of Physics at Harvard University and senior scientist at the Smithsonian Astrophysical Observatory. He is currently visiting professor at the University of Texas at Austin. He did his undergraduate work at Cornell University, being graduated in 1954, studied for the next year at the Niels Bohr Institute in Copenhagen and received his Ph.D. from Princeton University in 1957. Thereafter he worked at Columbia University, the Lawrence Radiation Laboratory of the University of California, the University of California at Berkeley and the Massachusetts Institute of Technology before going to Harvard in 1973. He has written two books: *The First Three Minutes: A Modern View of the Origin of the Universe* and *Gravitation and Cosmology: Principles and Applications of the General Theory of Relativity*. He has received numerous honors for his work on the theory of elementary particles, including five honorary degrees and the 1979 Nobel prize in physics, which he shared with Sheldon Lee Glashow and Abdus Salam. Weinberg was recently elected a foreign member of the Royal Society.

E. P. ABRAHAM ("The Beta-Lactam Antibiotics") is professor emeritus of chemical pathology at the Sir William Dunn School of Pathology of the University of Oxford. He did his undergraduate work at Oxford and obtained his D.Phil. there in 1938. After two years as a Rockefeller Foundation traveling fellow in Stockholm he returned to Oxford to work on the isolation and chemistry of penicillin with Howard W. Florey, Ernst B. Chain and others. In 1940 he and Chain discovered the enzyme penicillinase; in 1953 he and G. G. F. Newton isolated cephalosporin *C* from an impure preparation of penicillin N. Subsequent work at Oxford and in pharmaceutical companies led to the introduction of cephalosporins into medicine. Abraham, a Fellow of the Royal Society, was knighted last year. His leisure interests include walking and skiing.

CARL R. WOESE ("Archaebacteria") is professor of microbiology and of genetics and development at the University of Illinois at Urbana-Champaign; from 1972 to 1979 he held a third appointment as professor of biophysics. His bachelor's degree (in mathematics and physics) was awarded by Amherst College in 1950 and his Ph.D. (in biophysics) by Yale University in 1953. He then spent two years at the University of Rochester School of Medicine and Dentistry before returning to Yale to do research in biophysics. From 1960 to 1963 he worked as a biophysicist at the General Electric Research Laboratory, joining the faculty of the University of Illinois in 1964. "My entire career," he writes, "has been a deepening venture into the recesses of evolution." Much of his work has been on the evolution of the mechanism whereby the genetic code is translated in the cell by the ribosomes. Now, Woese says, "it is time to press deeper, and my interest is turning to the evolution of the ribosome itself."

ROBERT G. BLAND ("The Allocation of Resources by Linear Programming") is assistant professor in the School of Operations Research and Industrial Engineering and the Center for Applied Mathematics at Cornell University. "I studied at Cornell," he writes, "and got my Ph.D. in operations research there in 1974. I was assistant professor of mathematical sciences at the State University of New York at Binghamton, research fellow at the Center for Operations Research and Econometrics in Louvain and professor of management at the European Institute for Advanced Studies in Management in Brussels before returning to Cornell in 1978. My research interests are in the theory and applications of graphs and networks, mathematical programming and discrete optimization."

BERND HEINRICH ("The Regulation of Temperature in the Honeybee Swarm") is professor of zoology at the University of Vermont. He writes: "I grew up in rural Maine after coming to this country (when I was 10 years old) from Germany. I received bachelor's and master's degrees at the University of Maine and a Ph.D. from the University of California at Los Angeles, all in zoology. For the next 10 years I was in the entomology department at the Uni-